Safety Analyses for Power Uprate of VVER-1000/320 Temelín

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VVER – 2013, Experience and perspectives after Fukushima, Praha, Czech Republic,
11-13 November, 2013

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Introduction

- In the Czech Republic, there are 4 WWER-440 units and 2 WWER-1000 units in operation.
- At present, one of the current problems is feasibility of power uprate of these nuclear power plants.
- Specifically considered is the possibility to increase the core heat output by 3 – 9 %.
- The actual proposal is an increase of the core heat rate by 4 %, which corresponds to the 104 % of the nominal power (NPP Temelin, VVER 1000/320).

Introduction

- **Obviously, after the necessary changes, it is requisite to demonstrate that thus modified nuclear power plant is safe. Issuance of the subsequent new operation license is contingent on the results of the Safety Report revision.**
- **The paper presents a proposal of the power uprate of our nuclear power plant Temelín with WWER-1000/320 reactor and describes possible changes of the plant basic parameters. Discussion of these parameters impact on the method applied for the safety analyses performance within Chapter 15 (Safety Analyses) follows.**

Introduction

- **Proposed is also a procedure applied for the selection of limiting initiating events and then the actual solution. Briefly is evaluated possibility to apply the Best Estimate approach, taking into account uncertainties of the input data as well as that of the computer codes used.**

The following events are processed in the Safety Report Chapters

The following events are processed in the Safety Report Chapters:
15.1 Increase of heat removal by secondary circuit
15.2 Reduction of heat removal by secondary circuit
15.3 Reduction of coolant flow through primary circuit
15.4 RIA
15.5 Increase of mass of reactor coolant
15.6 Reduction of mass of reactor coolant
15.8 Anticipated transients without scram (ATWS)

ACCEPTANCE CRITERIA 1

Event analyses for part 15 of SAR are carried out in accordance with the requirements of the Czech Republic regulations and normative documentation of the Russian Federation, the requirements arising from the laws of the United States and the IAEA documents.

The criteria applied in analyses of representative initiating events determine requirements to fuel and to pressure limit in the primary and secondary circuits. Individual acceptance criteria are as follows:

ACCETAPNCE CRITERIA 2

ACCEPTANCE CRITERIA FOR TRANSIENTS

(1) **The probability of a boiling crisis anywhere in the core is low. This criterion is typically expressed by the requirement that there is a 95% probability at the 95% confidence level that the fuel rod does not experience a departure from nucleate boiling (DNBR).The DNBR correlation used in the analysis needs to be based on experimental data that are relevant to the particular core cooling conditions and fuel design.**

This acceptance criterion is met if minimum

DNBR > 1,348 with CRT-1 correlation for TVSA-T fuel.

(2) **The pressure in the reactor coolant and main steam systems is maintained below a prescribed value (typically 110% of the design pressure).**
Limit value of the primary pressure: 19,4 MPa.
Limit value of the secondary pressure: 8,69 MPa

ACCETAPNCE CRITERIA 3

ACCEPTANCE CRITERIA FOR TRANSIENTS

(3) **There is no fuel melting anywhere in the core.**
Fuel temperature shall be lower than the melting temperature:
In safety analysis the minimum values of the melting temperature of fuel rod and U-GD fuel rod (2840 °C and 2405 °C) are accepted that corresponds to maximum values of the burn up fuel burn up in tablet with provision for engineering factors.

ACCEPTANCE CRITERIA 4

In addition to criteria, particularly for design basis LOCAs, short term and long term core coolability should be ensured by fulfilling the following five criteria:

(4) The fuel rod cladding temperature should not exceed a prescribed value (typically 1200°C); the value is limiting from the point of view of cladding integrity following its quenching and is also important for avoiding a strong cladding–steam reaction, thus replacing criterion which is valid for other accidents.

(5) The maximum local cladding oxidation should not exceed a prescribed value (typically 17–18% of the initial cladding thickness before oxidation).

ACCEPTANCE CRITERIA 5

(6) The total amount of hydrogen generated from the chemical reaction of the cladding with water or steam should not exceed a prescribed value (typically 1% of the hypothetical amount that would be generated if all the cladding in the core were to react).

(7) Calculated changes in core geometry have to be limited in such a way that the core remains amenable to long term cooling, and the CRs need to remain movable.

(8) There should be sufficient coolant inventory for long term cooling.

ACCEPTANCE CRITERIA 6

(9) The radially averaged fuel pellet enthalpy does not exceed the prescribed values (the values differ significantly among different reactor designs and depend also on fuel burnup) at any axial location of any fuel rod.

This criterion ensures that fuel integrity is maintained and energetic fuel dispersion into the coolant will not occur (specific to RIAs).

ACCEPTANCE CRITERIA 7

The pressure in the reactor coolant and in the main steam system is maintained below a prescribed value (typically 135% of the design value for ATWSs and 110% for other DBAs).

This criterion ensures that the structural integrity of the reactor coolant boundary is maintained.

Calculated doses are below the limits for DBAs, assuming an event generated iodine spike and an equilibrium iodine concentration for continued power operation, and considering actual operational limits and conditions for the primary and secondary coolant activity.

ACCEPTANCE CRITERIA FOR ALL ACCIDENTS LEADING TO CONTAINMENT PRESSURIZATION 8

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In addition to the relevant criteria given above, the following criteria apply:

The calculated peak containment pressure needs to be lower than the containment design pressure and the calculated minimum containment pressure needs to be higher than the corresponding acceptable value.

Differential pressures, acting on containment internal structures important for containment integrity, have to be maintained at acceptable values.

Computer codes in Licensing Process

Computer code	Type of computer code	TH Models	Mather organisation	Suitable for ETE
ATHLET 3.0A	System program	1D TH / point neutron kinetics	SRN / GRS / GRS	yes
RELAP5	System program	1D TH / point neutron kinetics	USA / INEEL / US NRC	yes
RELAP5-3D	System program	3D TH+3D n. k.	USA / INEEL / US DOE	yes
DYN3D	TH -AZ and 3D neutr.kin	1D TH +3D n. k.	SRN / FzR/FzR	yes
ATHLET-DYN3D	System program+TH -3D neutr.kin	1D TH+3D n. k.	SRN / FzR/FzR SRN / GRS / GRS	yes
VIPRE 01	Subchanel	TH core and fuel assembly DNBR analyses	EPRI	yes
MELCOR	Containment	1D TH	USA / SandiaNL / US NRC	yes
FLUENT	CFD	3D TH	USA	yes
NEWMIX A REMIX	Mixing in RPV		USA / ? / US NRC	yes
COCOSYS	Containment	1D TH	SRN / GRS / GRS	yes

Main parameters

Parameters	Values and uncertaties			
	4 Loops		3 Loops	2 Loops
	101%	104%		
Reactor power, MW Uncertanties, %N_{nom}	3030 4	3120 4	64% 4	48% 4
Reactor coolant mass flow, м^3/hour: Min. Nominal Max.	82000 87500 91000	83200 88000 91000	59844[#] 65300 68800[#]	38032[#] 42300 45300[#]
Core inlet temperature., °C: Nominal	289,40	290,0	284.5 287.7-293.5 296	285.5 288.2-293.5 296
Core pressure, top.– abs., MPa*	15,7±0, 36	15,7± 0,36	15,7±0,36	15,7±0,36
Presurre, MSH– abs., MPa**	5.72 - 6,38		5.72 - 6,93	5.72 - 6,84
Core, Bypass ., %	3,5		3,5	3,5
PRZ level HFP, м* PRZ level HZP, м*	8,17***±10% 4,96±10%		8,17±10% 4,96±10%	8,17±10% 4,96±10%
SG Level, м****	2,36± 0.17	2,36±0. 17	2,36±0.17	2,36±0.17
SG Feed water temperature, °C*	220±5		196±5	196±5

Reactor trip system - FΔH

FΔH Remains the same

-> The absolute value of hot pin is higher

-> New CHF analyses, new computer codes

-> New core limits

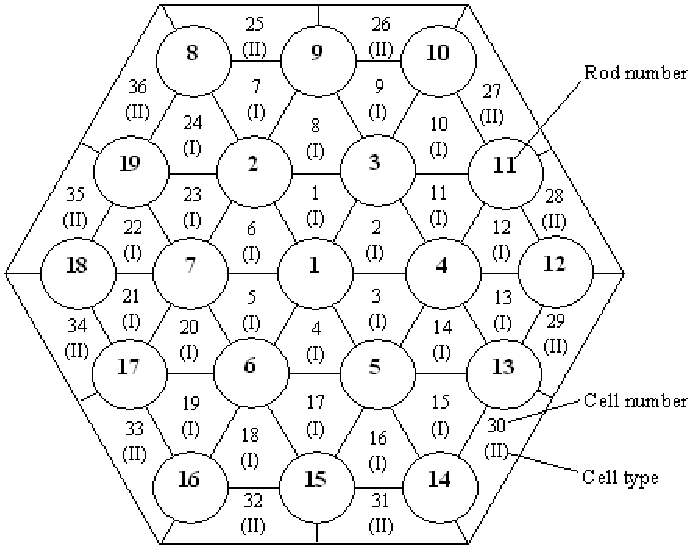
No.	TRIP	PRPS Reactor Trips			
		Group 1		Group 2	
		104%	100%	104 %	100%
1.	High Neutron Flux:				
	- Power Range- High Setting				
	4 MCP			> 108%	> 109%
4.	Overtemperature OTΔT	Core limits	Core limits		
7.	Power-to-Flow	Core limits	Core limits		
8.	Overpower OPΔT				
	- 4 MCP	>108%	>109 %		
9.	PRZ Low Pressure	<13,3 MPa	<12,0 MPa		

Temelin power uprate – use of VIPRE-01 and COURSE

- Use of VIPRE-01 in ÚJV Řež, a. s. in the frame of Temelin power uprate project:
  - To determine safety limits (DNBR) – VIPRE-01 + TVSA-T + CRT-1
  - To determine uncertainty of DNBR calculation – ΔDNBR
  - To calculate core limits
  - To calculate several safety analyses (LOFA, RIA) – subchannel code VIPRE
  - To calculate other IU – simple conservative code COURSE with isolated channel

Temelin power uprate – use of VIPRE-01 Experiments with TVSA-T

- **VIPRE-01 model:**



- **Statistical evaluation of results (95/95 approach) => safety limits for safety analysis.**

Temelin power uprate – use of VIPRE-01 and COURSE

Results:

	VIPRE	COURSE
Correlation limits	1.276	1.346

Increase in heat removal by the secondary system

No.	IU Chap. 15 of SAR	Results of conservative calculations				
		Primary pressure	Sec. pressure	DNBR	Cl. Temp.	Fuel temp.
15.1	Increase in heat removal by the secondary system	19,4 MPa	8,69 MPa	1,346 Course	1200 °C	2840 °C
15.1.5	Spectrum of steam system piping failures inside or outside the containment	HFP: HZP: MKV:	HFP: ZP: 7,5 MKV:	HFP: 1,711 HZP: 1,43 MKV: 1,866	HFP: 351 HZP: 331 MKV: 317,8	HFP: HZP: 2357 MKV: 1530,5

Decrease in heat removal by the secondary system

No.	IU Chap. 15 of SAR	Results of conservative calculations				
		Primary pressure	Sec. pressure	DNBR	Cladd Temp.	Fuel temp.
15.2	Decrease in heat removal by the secondary system, Limits	19,4 MPa	8,69 MPa	1,346 course	1200 °C	2840 °C
15.2.1	Turbine trip (closing of TG stop valves)	19,26	8,63	1,581		
15.2.4	Inadvertent closure of main steam isolation valves	19,33	8,38	1,559		
15.2.6	Loss of normal feedwater flow	19,36	8,34	1,357	365,3	

Decrease in heat removal by the secondary system 100% - 104%

Chapter	The initiation event	Analysis on the	Result/limit value for 104%	Result for 100%
15.2 Decrease in heat removal by the secondary system	15.2.1 Turbine trip (closing of TG stop valves)	DNBR	1,581/1.348	1,604
		Pressure PC.	18,81 MPa/19.4 MPa	16.81 MPa
		Pressure , SC	8.63 MPa/ 8.69 MPa	8.52 MPa
	15.2.6 Loss of normal feedwater flow	DNBR PRESSURE	1,357/1.348 19,36 MPa/19.4 MPa	1,642 18,5 MPa

Decrease in reactor coolant system flow rate

No.	IU Chap. 15 of SAR NPP Temelín	Results of conservative calculations				
		Prim. pressure	Sec. pressure	DNBR	Cladd. Temp.	Fuel temp.
15.3	Decrease in reactor coolant system flow rate, Limits	19,4 MPa	8,69 MPa	1.276_{VIPRE}	1200. °C	2840 °C
15.3.2	Sequential loss of forced reactor coolant flow			1,297	initial value + 2	
15.3.3	Complete loss of forced reactor coolant flow (all MCP trips)			4MCP: 1,499		
15.3.4	MCP shaft seizure (locked rotor)	18,49	8,42		765 °C	

Decrease in reactor coolant system flow rate 100% - 104%

Chapter	The initiation event	Analysis on the	Result/limit value for 104%	Result for 100%
15.3 Decrease in reactor coolant system flow rate	15.3.1 Single and multiple MCP trips 1 of 4 MCP 2 of 4 MCP 1 of 3 MCP	DNBR DNBR DNBR	1.502/1.276 1.516/1.276 1.602/1.276	– 1,587/1.348 –
	15.3.2 Sequential loss of forced reactor coolant flow 1 + 3 HCČ 2 + 2 HCČ	DNBR DNBR	1.297/1.276 1,363/1.276	1,421/1.348 1,575/1.348
	15.3.3 Complete loss of forced reactor coolant flow (all MCP trips) 4 of 4 MCP	DNBR	1,499/1.276	1,567/1.348

Reactivity and power distribution anomalies (RIA) 1

Increase in reactor coolant inventory, limits

Decrease in reactor coolant inventory

IE	Scenario according to Chap. 15 of SAR	Results of conservative calculations				
		Prim. pressure	Sec.dary pressure	DNBR	Cladd. Temp.	Fuel temp.
15.6	Decrease in reactor coolant inventory	19,4 MPa	8,69 MPa	*Lim 1,346*	1200 °C	2840 °C
15.6.1	Inadvertent opening of a pressurizer safety or relief valve		8,34	1,357		
15.6.4	Loss-of-coolant accident (LOCAs) (small break)				714°C	
15.6.5	Loss-of-coolant accident (LOCAs) (large break)				1045°C	

Decrease in reactor coolant inventory 100% - 104%

Chapter	The initiation event	Analysis on the	Result/limit value for 104%	Result for 100%
15.6 Decrease in reactor coolant inventory	15.6.1 Inadvertent opening of a pressurizer safety or relief valve	DNBR	1,375/1.348	1, 42/1.28
	15.6.5 Loss-of-coolant accident (LOCAs) (small break)	Cladding temperature	714 ° c/1200 ° c	652°c/1200 °c
	15.6.6 Loss-of-coolant accident (LOCAs) (large break)	Cladding temperature	1045° C	1045° C

Methodology of analyses

- The conservative and best estimate approaches have been used in most countries, even though regulatory bodies in different countries have tailored these approaches to fit their particular needs.

- Present regulations permit the use of best estimate codes, but there may be added requirements for conservative input assumptions, sensitivity studies or uncertainty studies.

Methodology of Analyses:

- Brief description and selection of methodology for uncertainty and sensitivity analyses.
- Description of uncertainty methods and philosophy of their selection.
- Examples of use

Conservative versus best estimate approach

Applied codes	Applied codes Input & BIC (boundary and initial conditions)	Assumptions on systems availability	Approach
Conservative codes	Conservative input	Conservative assumptions	Deterministic
Best estimate (realistic) codes	Conservative input	Conservative assumptions	Deterministic
Best estimate codes + Uncertainty	Realistic input + Uncertainty	Conservative assumptions	Deterministic
Best estimate codes + Uncertainty	Realistic input + Uncertainty	PSA-based assumptions	Deterministic + Probabilistic

A conservative approach

does not give any indication:

- about actual plant behaviour,
- including timescale,
- for preparation of EOPs or
- for use in accident management and
- preparation of operation manuals
- for abnormal operating conditions.

Selection of an uncertainty method for solution of a particular task

Was based on comparison of all monitored methods.

We come to the conclusion that the most suitable will be the nonparametric method based on Wilk's Formula

(GRS, IRSN).

Examples of BE-GRS methodology
The use of BE-access the Best Estimate

The methodology of the Best Estimate approach for SA processed events:

- LB LOCA
- SB LOCA
- PRISE
- Seizure of the rotor of MCP
- Loos of flow
- MSLB

LB LOCA - Parameters of the fuel pins

Properties of the fuel pins, and the parameters for the calculation of the conductivity of the gas gap (fuel – clading) shall be specified in accordance with the design data of the fuel, on the basis for various values of burnout.

For the calculation of the conductivity of the gas gap model from ATHLET was used.

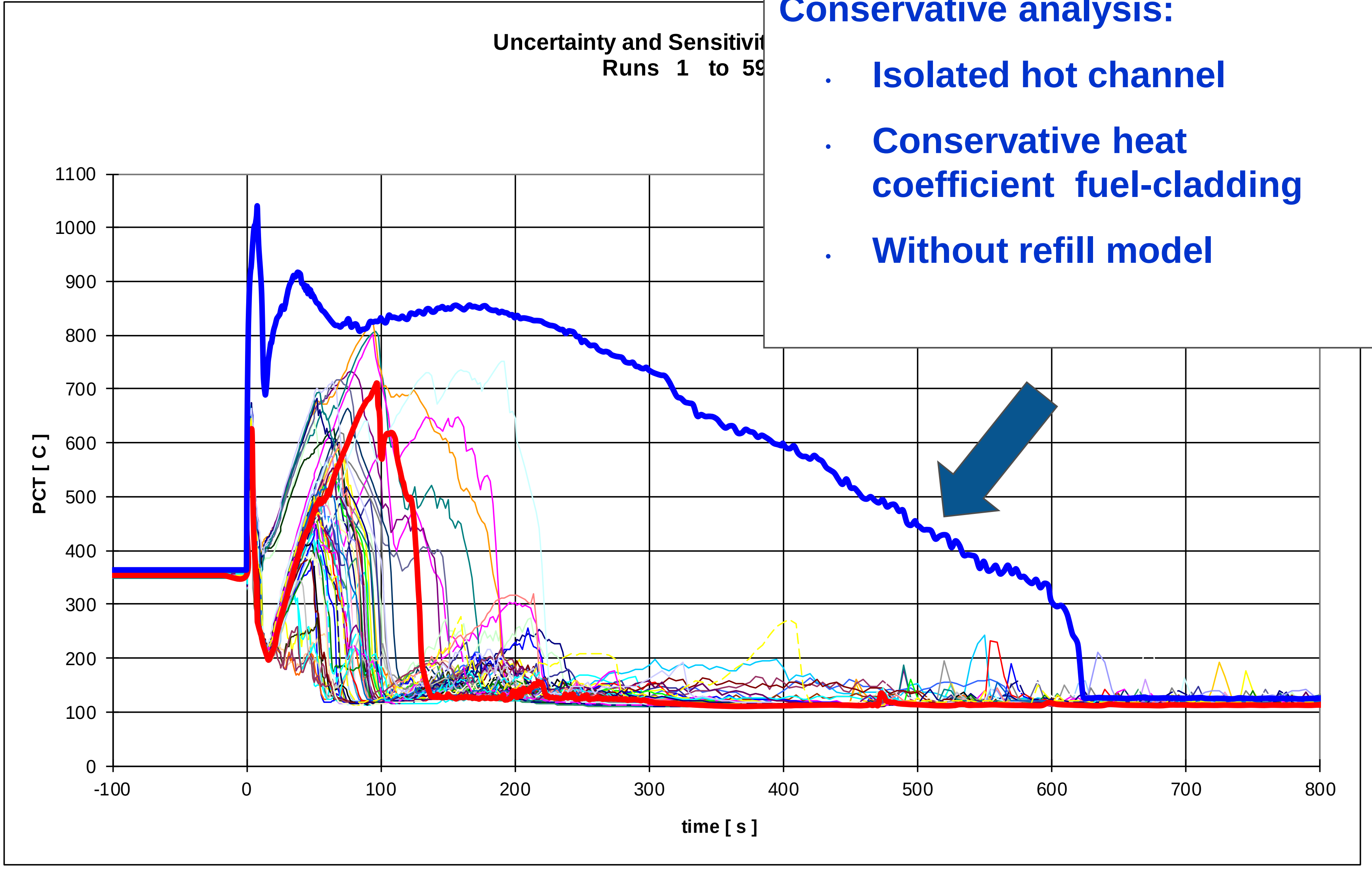
This conductivity is most important parameter for LOCA (PCT)

Important parameters of LB LOCA analysis.

Gap model, core nodalization

Parameter	Conservative calculation	Best estimate calculation
Heat Transfer Gap fuel - clad	Constant, minimum	Model
Core nodalization	Isolated channel	Cross flow between fuel assemblies
Refill model	No	Yes

The Results of the LOCA Analysis – max. PCT



Conservative analysis:

- **Isolated hot channel**
- **Conservative heat coefficient fuel-cladding**
- **Without refill model**

Comments on LOCA

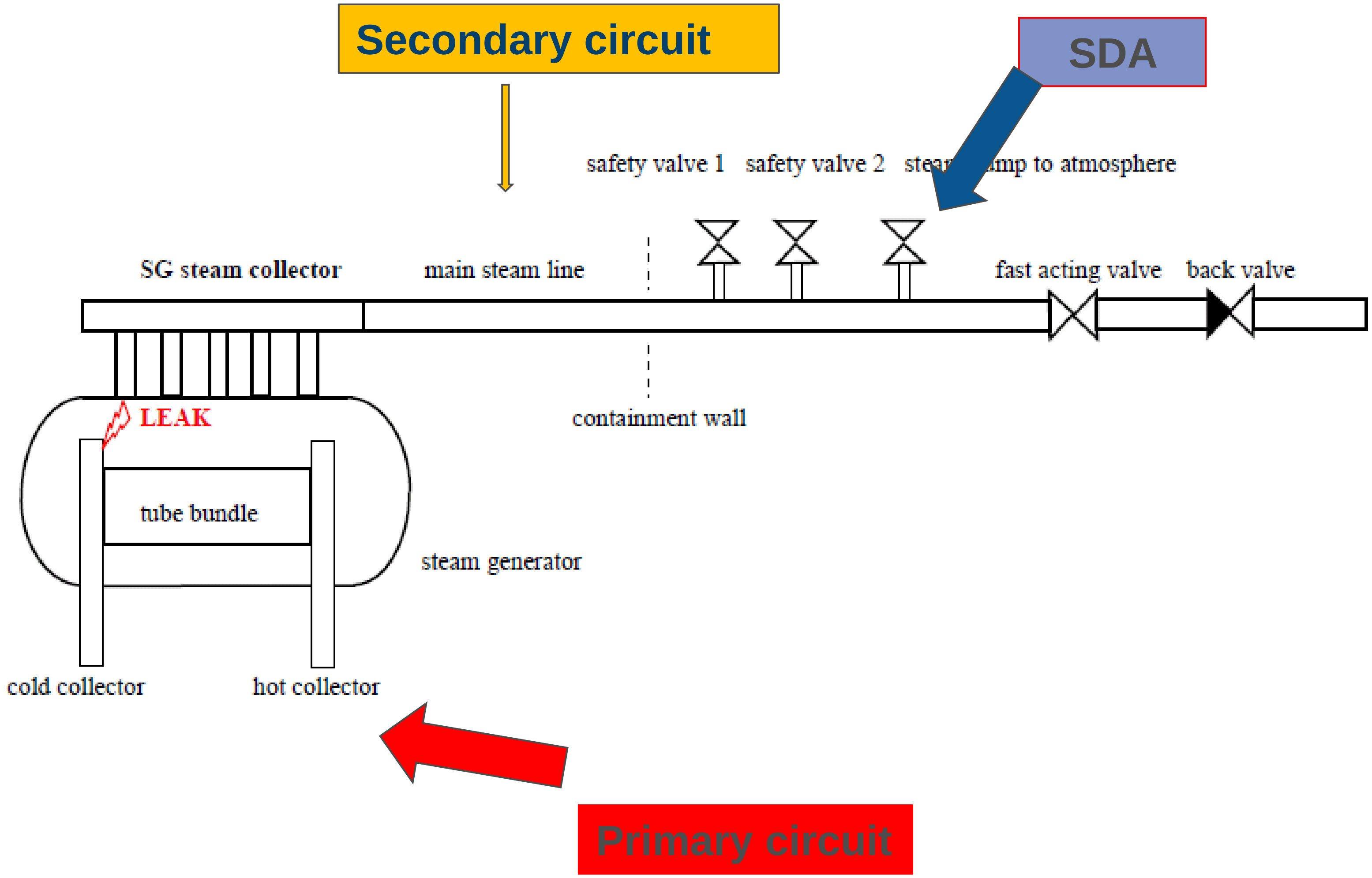
The methodology is applied to a specific events of SAR

Results are input for thermo mechanical analyses and for analysis of containment

The methodology is a qualitatively new step in safety analysis

The results of the analysis are significantly more favourable than the conservative analysis

PRISE-Analysis



The output parameters of analysis

- Maximum fuel cladding temperature
- Pressure in primary circuit
- Total mass in the primary circuit
- Break mass flow rate, primary-secondary circuit
- Integral break mass to atmosphere

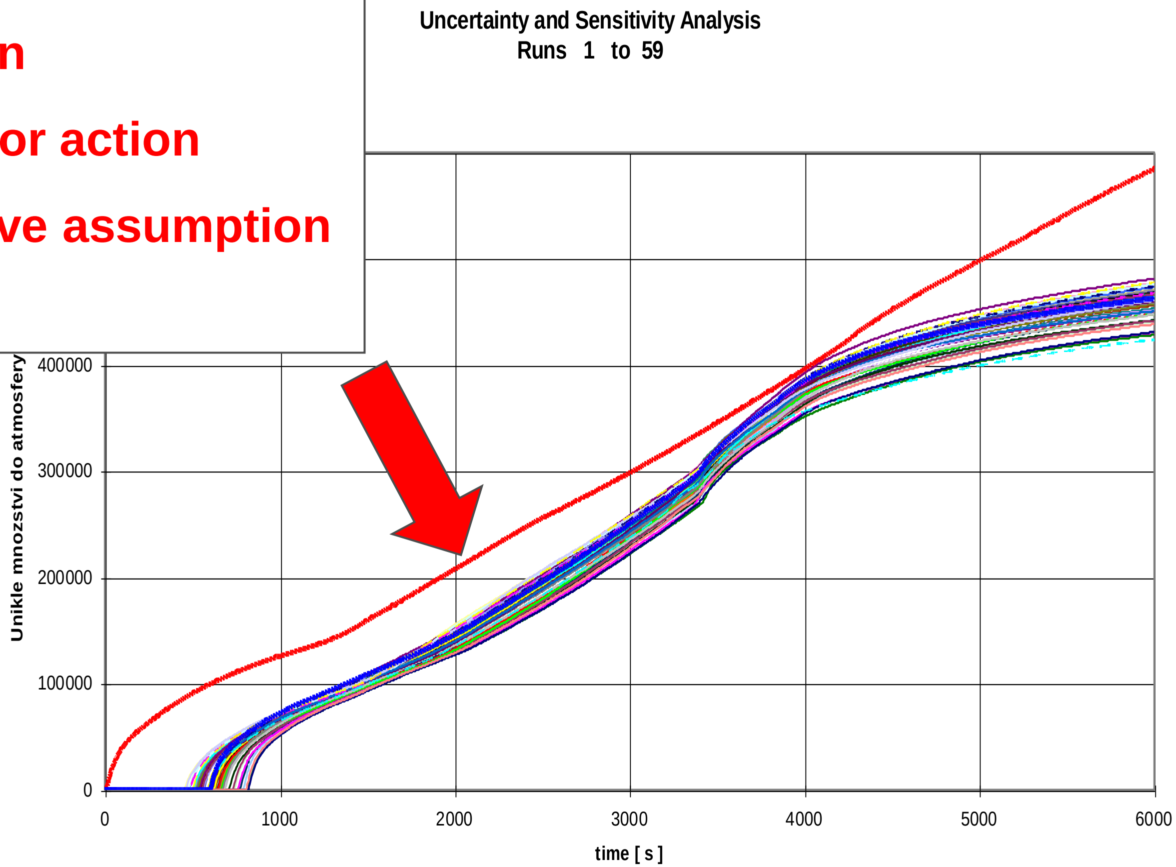
Selection of the computer code

Advanced best-estimate TH code ATHLET or RELAP

PRISE NPP, comparison BE and coservative analysis

Conservative analysis:

- **SDA stuck open**
- **Without operator action**
- **BIC conservative assumption**



Integral mass release to atmosphere (SDA). SDA stuck open.

Comments on PRISE Analysis

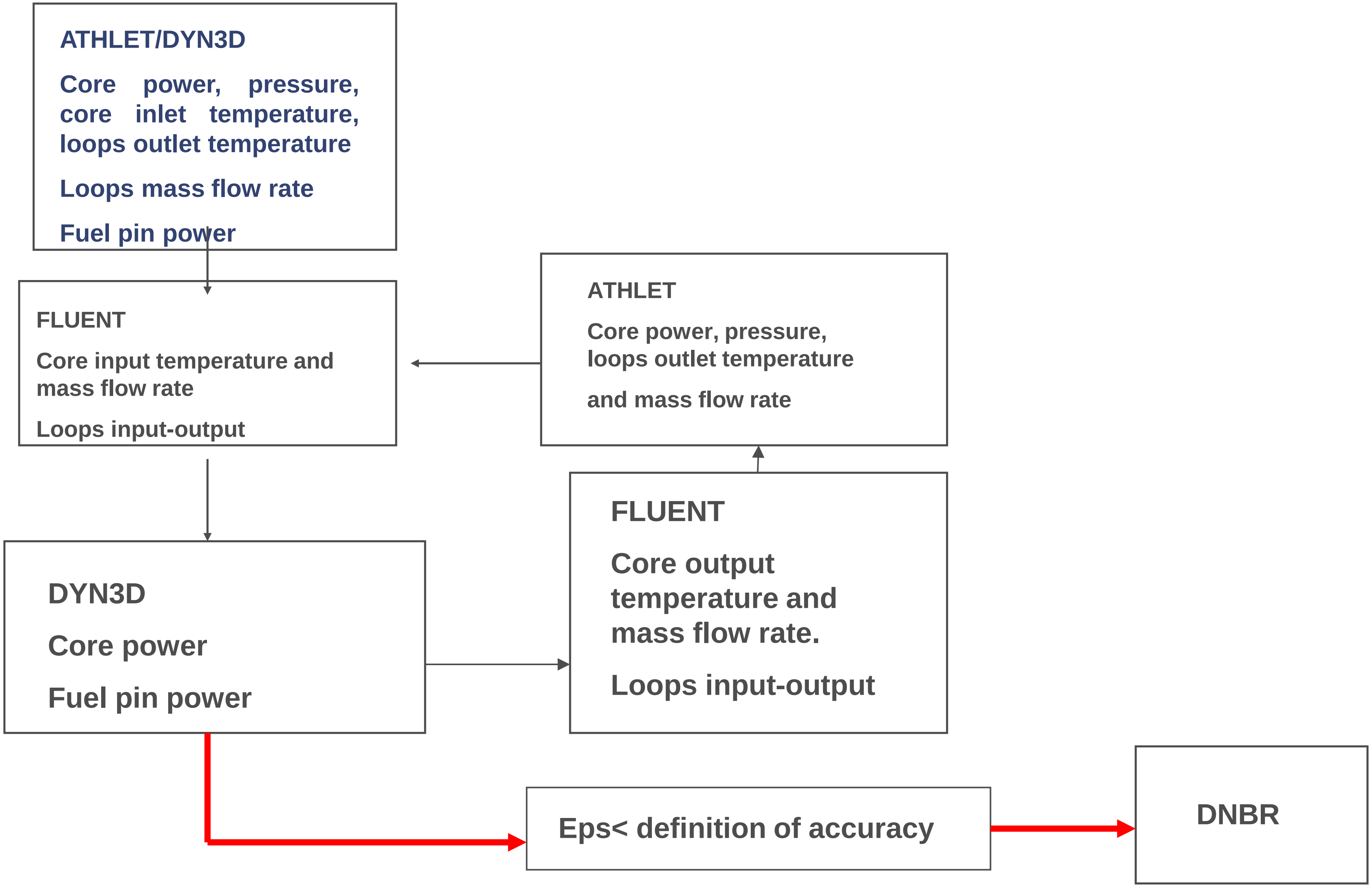
Difference between the amount of leaked mass into the atmosphere - the influence of the radiological consequences

MSLB WWER – 1000/320 Temelín. Scope of analyses.

Focused on DNBR determination.

Calculations were performed with coupled version of ATHLET/DYN3D code for the unit under hot zero power conditions at the end of fuel cycle, with reactivity coefficients corresponding to the project limits and different number of MCPs in operation.

MSLB VVER – 1000. Schema of the Analyses.



Selected 21 uncertain input parameters:

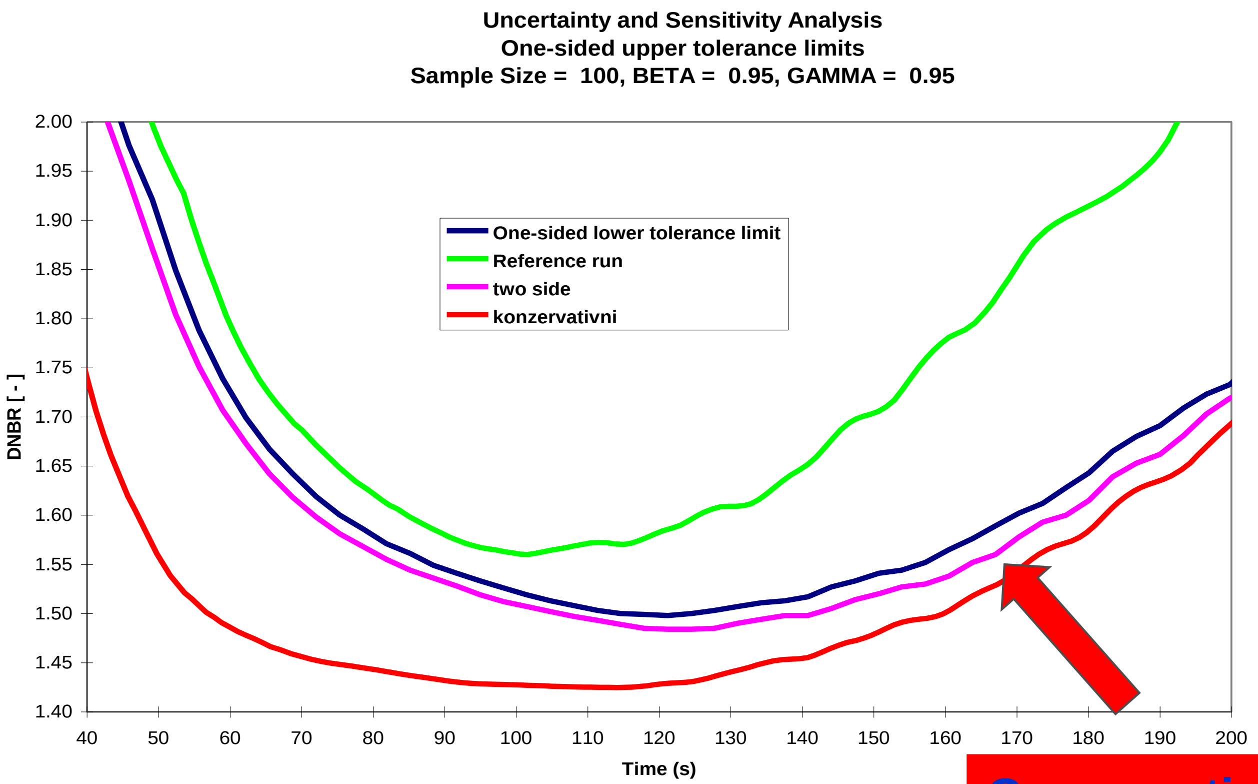
Models of:

- Critical break flow
- Reactivity coefficients
- Boundary and initial conditions
- Reactor power
- HPI System parameters
- Boron concentration
- Feed water parameters
- Emergency feed water parameters
- Control system parameters

Major initial conditions.

Parameter	Conservative calculation	Best Estimate calculation
Decay heat	Zero	ANSI/ANS-5.1-1979 (- 20 %)
PRZ level	Minimal	Design value for HZP
Primary pressure	Maximal	Design value
Reactor flow	Minimal	Design value
Inlet temperature	Maximal	Design value for HZP
Secondary pressure	Maximal (in order to get maximal primary temperature)	Design value

MSLB - DNBR Analysis.



Conservative analysis

Comparison of the results to BE a conservative analysis

DNBR

- Best estimate approach was calculated the minimum value DNBR 1.826
- The DNBR correlation limit is 1.348, a minimum margin for BE is 36 %.
- A conservative calculation of the minimum value has been reached DNBR 1,43. Minimum margin is 7%.
- Difference between the conservative and BE approach is 30 %.

The failure of MCP, rationale for selecting particular initial events

- Failure 1 from 4s working MCP, with consequent failure of the remaining MCP was, in the case of conservative analysis, worst initial events in terms of DNBR.

- The correlation limit for VIPRE code is 1,276 (subchannel analysis).

- RELAP5 and VIPRE- 01 programs were used for the calculations

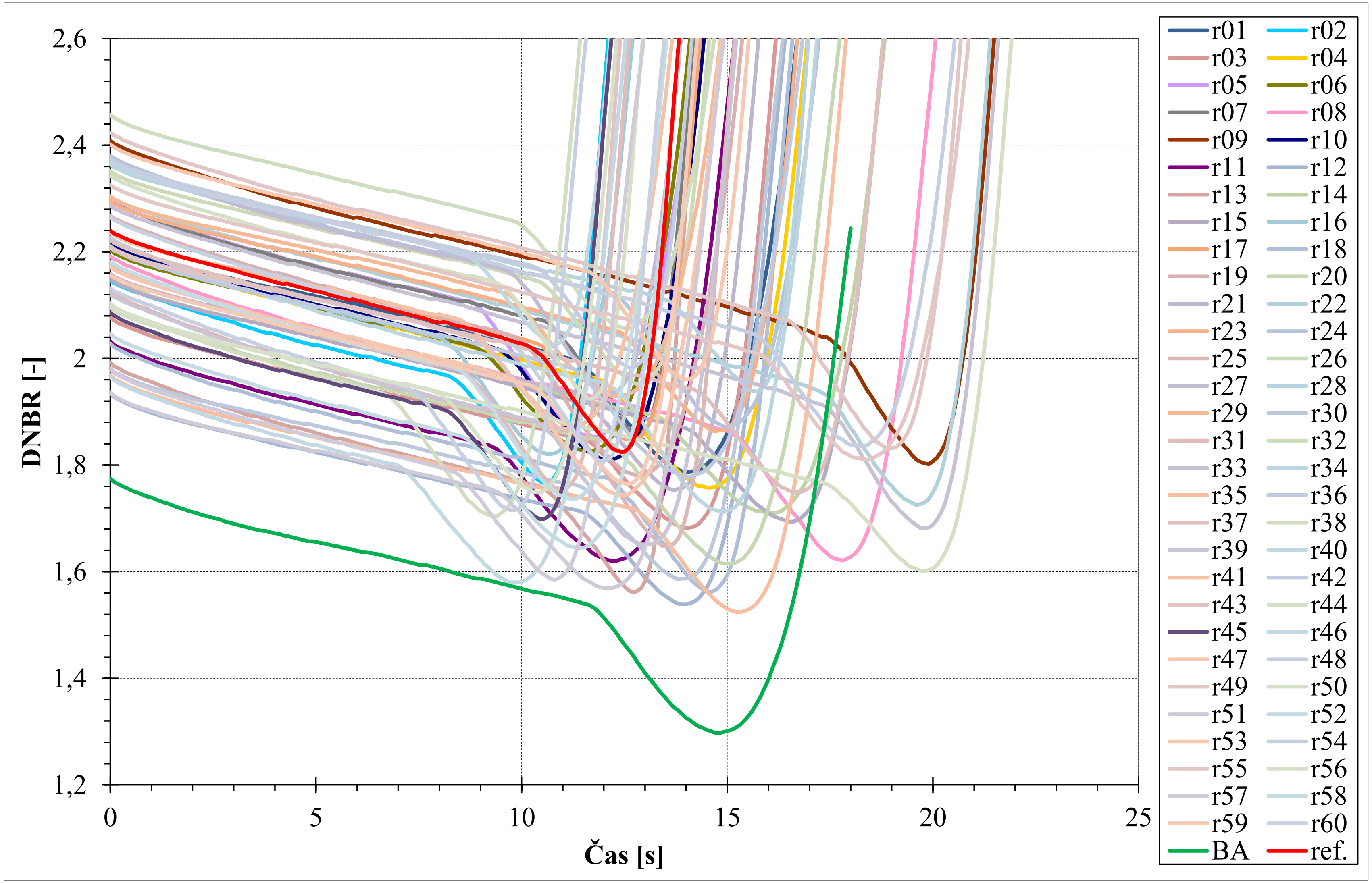
Selection of uncertainties input parameters and models

From the set of initial parameters were chosen 11 most important

PAR.	PARAMETR	Unit	Initial value	Uncertainty range	PDF
1	Relative value of initial reactor power	–	1,0	±0,04	Uniform
2	Decay heat (multiplier)	–	1,0	±0,15	Normal
3	The flow of the coolant at the entrance to the reactor	m^3/hr	88000	83000 ÷ 91000	Uniform
4	The pressure in the Pressurizer	MPa	15,7	±0,36	Uniform
5	The water level in the Pressurizer	m	8,36	±10 %	Uniform
6	The pressure in the main steam colector	MPa	6,08	5,76 ÷ 6,42	Uniform
7	Fuel Temperature coefficient of reactivity	1/°C	$-2{,}2\cdot10^{-5}$	$-4{,}2\cdot10^{-5} \div -1{,}6\cdot10^{-5}$	Uniform
8	Density coefficient of reactivity	$1/(g/cm^3)$	0,1	0,03 ÷ 0,19	Uniform
9	Control rods delay	s	1,75	1,5 ÷ 4,3	Uniform
10	The heat transfer coefficien of the gap fuel-clading	$W/(m^2\cdot K)$	20000	6352,6 ÷ 33468	Uniform
11	Reactor Trip Signal settings	%	0	-5,9 ÷ 4,0	Uniform

Srovnání výsledků BE a konzervativní analýzy

DNBR green - conservative analysis, red =reference analysis, 59 runs



Comments on Loss of Flow

- The acceptability criterion is met. The minimum value of DNBR was 1,525 (1,276 limit). In the case of a conservative calculation of the minimum was 1,297. The reserve to acceptance criterion is 19.5% compared to 1.6%.

SUMMARY OF THE RESULTS OF THE SAFETY ANALYSIS

- Analyses of the accidents were assembled in accordance with the requirements of the Czech Republic and were based on the philosophy of the representative (bounding) safety analysis.

- The analyzed results represent limiting cases for each of the initiation event. For all the analysis of processes ANSI category II is the calculated minimum DNBR larger than the relevant limit value.

- High pressures of the RCS and MSS remain below the safety limits

SUMMARY OF THE RESULTS OF THE SAFETY ANALYSIS

- For ANSI event category III the applicable criterion of acceptability is specified for each event. The results for each subject category III event meet the specified criteria.

- For ANSI Event Category IV the applicable criterion of acceptability is specified for each event. The results for each subject category IV event meet the specified criteria.

- Also were presented as an independent analysis of the selected event, conducted initiation so called best estimate method – the Best Estimate (BE).

Thank you for your attention